U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.5.1 – 15.5.2 INADVERTENT OPERATION OF ECCS AND CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch ~~(RSB)~~(SRXB)[1]

Secondary - None

I. AREAS OF REVIEW

Various types of equipment malfunctions, operator errors, and abnormal occurrences that may occur with moderate frequency[12] can cause an unplanned increase in reactor coolant inventory. Depending on the boron concentration and temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could[3] lead to fuel damage or overpressurization of the reactor coolant system. Alternatively, a power level decrease and depressurization may result. The reactor will trip from high water level, high flux, or high or low pressure.

This Standard Review Plan (SRP)[4] section is intended to be applicable to~~ these types of~~[5] moderate frequency events that increase reactor coolant inventory. These transients should be discussed in individual sections of the applicant's safety analysis report (SAR), as required by ~~the~~

[1] The term "moderate frequency" is used in this SRP section in the same sense as it is used in the description of design and plant process conditions in Reference 8 and as "frequent" is used in Reference 9.

~~Standard Format (Ref. 1)~~Regulatory Guide 1.70.[6] The specific initiating events considered in SRP section are:

1. Boiling water reactors (BWRs) — Inadvertent operation of the high pressure core spray, high pressure coolant injection, or reactor core isolation cooling system.

2. Pressurized Water Reactors (PWRs)~~PWRs~~[7] — Inadvertent operation of high pressure emergency core cooling system (high pressure injection system) or a malfunction of the chemical and volume control system.

Other BWR transients that can result in an increase in reactor coolant inventory include feedwater system malfunctions (increasing flow), steam pressure regulator malfunctions (decreasing flow), loss of electrical load, turbine trip, main steam isolation valve (MSIV) closure, and loss of condenser vacuum. These transients are the subject of other SRP sections that consider their effects on system parameters other than coolant inventory. However, the impact of these transients on reactor coolant inventory is considered by the reviewer as a portion of the effort involved in this SRP section.

The review of events leading to an increase in reactor coolant inventory considers the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the transient.

The sequence of events described in the applicant's SAR is reviewed by ~~RSB~~SRXB.[8] The ~~RSB~~SRXB reviewer concentrates on the need for the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed by ~~RSB~~SRXB[9] to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the ~~RSB~~SRXB[10] reviewer initiates a generic evaluation of the new analytical model. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of those transients analyzed are reviewed to ~~assure~~ensure[11] that the consequences meet the acceptance criteria given in subsection II, below.

Further, the results of the analysis are reviewed to ascertain that the values of pertinent system parameters are within ranges expected for the type and class of reactor under review.

Review Interfaces[12]

The ~~RSB~~SRXB[13] will coordinate other branch evaluations that interface with the overall review of the transient analysis as follows:[14]

The Instrumentation and Controls~~ Systems~~ Branch ~~(ICSB)~~(HICB)[15] reviews the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems as part of its primary review responsibility for SRP Sections 7.2 through 7.5.

~~The Core Performance Branch (CPB), upon request from RSB, reviews the values of the parameters used in the analytical models which relate to the reactor core for conformance to plant design and specified operating conditions; determines the acceptance criteria for fuel cladding damage limits; and reviews the core physics, fuel design, and core thermal-hydraulics data used in the SAR analysis as part of its primary review responsibility for SRP Sections 4.2 through 4.4.~~[16]

~~The Accident Evaluation Branch (AEB), using fuel damage results provided by RSB, evaluates the radiological consequences associated with the fuel failure.~~[17]

The review of the Technical Specifications is coordinated and performed by the ~~Licensing Guidance Branch (LGB)~~Technical Specifications Branch (TSB)[18] as part of its primary review responsibility for SRP Section 16.0.

The SRXB also performs the following review under the SRP sections indicated:

The SRXB reviews the values of the parameters used in the analytical models which relate to the reactor core for conformance to plant design and specified operating conditions; determines the acceptance criteria for fuel cladding damage limits; and reviews the core physics, fuel design, and core thermal-hydraulics data used in the SAR analysis as part of its primary review responsibility for SRP Sections 4.2 through 4.4.[19]

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding review branch.

II. ACCEPTANCE CRITERIA

The ~~RSB~~SRXB[20] acceptance criteria are based on meeting the relevant requirements of the following regulations:

A. General Design Criterion 10 (GDC 10),[21] as it relates to the reactor coolant system being designed with appropriate margin to ~~assure~~ensure[22] that specified acceptable fuel design limits are not exceeded during normal operations, including anticipated operational occurrences.

B. General Design Criterion 15 (GDC 15),[23] as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to ~~assure~~ensure[24] that the pressure boundary will not be ~~breeched~~breached[25] during normal operations, including anticipated operational occurrences.

C. General Design Criterion 17 (GDC 17) as it relates to providing onsite and offsite electric power systems to ensure that structures, systems, and components important to safety will function during normal operation, including anticipated operational occurrences. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded during an anticipated operational occurrence.[26]

D~~C~~. General Design Criterion 26 (GDC 26),[27] as it relates to the reliable control of reactivity changes to ~~assure~~ensure[28] that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. This is accomplished by ~~assuring~~ensuring[29] that appropriate margin for malfunctions, such as stuck rods, ~~are~~is[30] accounted for.

The basic objectives in reviewing the events leading to an increase in reactor coolant inventory are:

1. To identify which of the moderate frequency events leading to a coolant inventory increase are the most limiting.

2. To verify that, for the most limiting transients, the plant responds to the core flow increase in such a way that the criteria regarding fuel damage and system pressure are met.

The specific criteria necessary to meet the requirements of ~~GDC~~ General Design Criteria[31] 10, 15, and 26 for incidents of moderate frequency are:

a. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design ~~valves~~values ~~(Ref. 2)~~in accordance with the ASME Boiler and Pressure Vessel Code.[32]

b. Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR)~~DNBR~~[33] remains above the 95/95 DNBR limit for PWRs and the ~~CPR~~ critical power ratio (CPR)[34] remains above the ~~MCPR~~minimum critical power ratio (MCPR)[35] safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

c. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

d. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered~~ and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations~~.[36] For such accidents, fuel failure must be assumed for all rods for which the DNBR or CPR falls below those ~~valves~~values[37] cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2) that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.

e. To meet the requirements of General Design Criteria 10, 15 and 26 the guidelines of Regulatory Guide 1.105, "~~Instrument Spans and Setpoints~~Instrument Setpoints for Safety-Related Systems,"[38] are used with regard to their impact on the plant response to the type of transient addressed in this SRP section.

f. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the guidelines stated in Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems." ~~(Ref. 14).~~[39]

The applicant's analysis of events leading to an increase of reactor coolant inventory should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References ~~5~~ 8, 10, and 12 through ~~8~~ 15[40] are acceptable. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer initiates an evaluation.

The values of parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model:

a. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

b. Conservative scram characteristics are assumed, i.e., for a PWR maximum time delay with the most reactive rod held out of the core and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate.

c. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

d. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with Regulatory Guide 1.105. Compliance with Regulatory Guide 1.105 is determined by ~~ICSB~~HICB under the SRP Chapter 7 reviews.[41]

<u>Technical Rationale</u>

The technical rationale for application of these acceptance criteria to reviewing the inadvertent operation of the ECCS and of the chemical and volume control system malfunctions is discussed in the following paragraphs:[42]

1. Compliance with GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to ensure that fuel design limits are not exceeded during anticipated operational occurrences.

   The requirements of GDC 10 apply to this section because the reviewer evaluates certain events occurring with moderate frequency and having the potential to result in exceeding fuel design limits. The reviewer evaluates the models for determining the values of the fuel design parameters during such accidents to ensure that the limits will not be exceeded.

   Meeting this criterion provides assurance that anticipated operational occurrences will not result in fuel damage and subsequent fission product release.[43]

2. Compliance with GDC 15 requires that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during anticipated operational occurrences.

   The requirements of GDC 15 apply to this section because the reviewer evaluates certain events occurring with moderate frequency and having the potential to cause pressure transients. The reviewer evaluates the models used by the applicant to determine resultant coolant pressures and determines that pressures will stay within limits that preclude damage to the pressure boundary.

   Meeting this criterion provides assurance that anticipated operational occurrences will not result in damage to the reactor coolant pressure boundary and subsequent fission product release.[44]

3. Compliance with GDC 17 requires that onsite and offsite electrical power systems be provided to ensure that structures, systems, and components important to safety will perform their intended function. Each power system (assuming the other system is not functioning) shall provide sufficient capacity and capability to ensure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences.

   GDC 17 is applicable to SRP Section 15.5.1-15.5.2 because this section reviews the analysis of a group of abnormal operating occurrences to which the GDC must be applied.

   Meeting the requirements of GDC 17 provides assurance that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of initiating events involving an increase in reactor coolant inventory, concurrent with a loss-of-offsite-power (LOOP).[45]

4. Compliance with GDC 26 requires that the reactivity control system utilizing control rods be capable of reliably controlling reactivity changes to ensure that, under anticipated

operational occurrences and with appropriate margin for malfunctions such as stuck rods, fuel design limits are not exceeded.

The requirements of GDC 26 apply to this section because the reviewer looks at certain events occurring with moderate frequency and having the potential to result in exceeding fuel design limits. The reviewer, in conjunction with the HICB reviewer implementing SRP Chapter 7, determines that the design parameters of the reactivity control system are sufficient to control the reactivity changes resulting from the anticipated operational occurrences.

Meeting this criterion provides assurance that anticipated operational occurrences will not result in fuel damage and subsequent fission product release.[46]

III. REVIEW PROCEDURE

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

The applicant's description of events leading to an increase in reactor coolant inventory is reviewed by ~~RSB~~SRXB[47] regarding the occurrences leading to the initiating event. The sequence of events, from initiation until a stabilized condition is reached, is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that is required.

5. The extent to which operator actions are required.

6. That appropriate margin for malfunctions, such as stuck rods (see II.3.b), ~~are~~is[48] accounted for.

The applicant should present a quantitative analysis in the SAR of the event leading to an increase in reactor coolant inventory which is the most limiting. For this event, the ~~RSB~~SRXB[49] reviewer, with the aid of the ~~ICSB~~HICB[50] reviewer, reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the event to acceptable levels. The ~~RSB~~SRXB[51] reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The ~~ICSB~~HICB[52] review of

Chapter 7 of the SAR confirms that the instrumentation and control systems design is consistent with the requirements for safety system actions for these events.

To the extent deemed necessary, the ~~RSB~~SRXB[53] reviewer evaluates the effects of single active failures of systems and components which may affect the course of the transient. For new applications, loss of offsite power (LOOP) should not be considered a single failure; each increase of inventory transient should be analyzed with and without a LOOP in combination with a single active failure. (This position is based upon interpretation of GDC 17, as documented in the Final Safety Evaluation Report for the ABB-CE System 80+ design certification.)[54] In this phase of the review, the system reviews are performed as described in the SRP sections for Chapters 4, 5, 6, 7, 8, and 9 of the SAR.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by ~~RSB~~SRXB[55] to determine if these models have been previously reviewed and found acceptable by the staff. If not, a generic review of the models is initiated.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by ~~RSB~~SRXB.[56] Of particular importance are the reactivity coefficients and control rod worths used ~~by the applicant in his analysis~~ in the applicant's analysis,[57] and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that ~~he has selected~~ the core burnup has been selected[58] that yields the minimum margins is evaluated. ~~CPB~~The appropriate SRXB reviewer[59] is consulted regarding the values of the reactivity parameters used in the applicant's analysis.

The results of the applicant's analysis are reviewed and compared ~~to~~ with[60] the acceptance criteria-presented in subsection II regarding maximum pressure in the reactor coolant and main steam systems and the minimum critical heat flux ratio (MCHFR) or ~~departure from nucleate boiling ratio (DNBR)~~DNBR.[61] The variations with time during the transient of the neutron power, heat fluxes (average and maximum), reactor coolant system pressure, minimum DNBR (PWR) or CPR (BWR); core and recirculation loop coolant flow rates (BWR), coolant conditions (inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam fractions),[62] steam line pressure, containment pressure, pressure relief valve flow rate, and flow rate from the reactor coolant system to the containment system (if applicable) are reviewed. The values of the more important of these parameters for the events leading to an increase in reactor coolant inventory are compared ~~to~~ with those predicted for other similar plants to confirm that they are within the expected range.

Note: In the Final Safety Evaluation Report for the Advanced Boiling Water Reactor (ABWR) (Reference 9), the staff allowed an exception to Acceptance Criterion II.A for the postulated downscale failure of steam pressure regulator. Normally, such transients are treated as anticipated operational occurrences, which must not result in specified acceptable fuel design limits being exceeded. For this special case, the transient, requiring coincident failure of three independent channels, is not expected to occur during the lifetime of the plant and is not classified as an anticipated operational occurrence, but rather as an anticipated transient involving a common-mode software failure. Accordingly, the following criterion for the radiological dose calculation was established: the resulting dose due to fuel failures should not exceed 10 percent of 10 CFR Part 100, which is considered appropriate for an event of such frequency because of the unique design features of ABWR instrumentation and control systems.[63]

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[64]

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and ~~his~~that the[65] review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report (SER):

> A number of plant transients can result in an increase in reactor coolant inventory. Those that might be expected to occur with moderate frequency are inadvertent operation of the emergency core cooling system, chemical and volume control system malfunction, and various BWR transients.** All these postulated transients have been reviewed. It was found that the most limiting in regard to ________________ was the ________________ transient.

The staff concludes that the analysis of a transient resulting in an unplanned increase in heat removal by the secondary system due to an increase in reactor coolant inventory is acceptable and meets the requirements of General Design Criteria 10, 17,[66] 15, and 26. This conclusion is based on the following:

1. In meeting ~~GDC~~General Design Criteria 10, 15, and 26 as indicated below we have determined that the applicant's analysis was performed using a mathematical model that had been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative. In addition, we have further determined that the positions of Regulatory Guide 1.53 for the single-failure criterion and Regulatory Guide 1.105 for instruments have also been satisfied.

2. The applicant has met the requirements of GDC 10, 17,[67] and 26 with respect to demonstrating that resultant fuel damage is maintained since the specified acceptable fuel design limits were not exceeded for this event.

3. The applicant has met the requirements of GDC 15 and 17[68] with respect to demonstrating that the reactor coolant pressure boundary limits have not been exceeded

** The SER draft should present one statement for all similar transients.

by this event and that resultant leakage will be within acceptable limits. This requirement has been met since the maximum pressure within the reactor coolant and main steam systems did not exceed 110% of the design pressures.

4. The applicant has met the requirements of GDC 26 with respect to the capability of the reactivity control system to provide adequate control of reactivity during this event while including appropriate margins for stuck rods since the specified acceptable fuel design limits were not exceeded.

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[69]

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[70] Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.[71]

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and NUREGs.

VI. REFERENCES

1. CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

2. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant System Design."

3. 10 CFR Part 50, Appendix A, General Design Criterion 17, "Electric Power Systems."[72]

4. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

5. Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."[73]

6~~1~~. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

7. Regulatory Guide 1.105, "Instrument Setpoints for Safety-Related Systems."[74]

8. NUREG-1503, "Final Safety Evaluation Report (FSER) Related to the Certification of the Advanced Boiling Water Reactor (ABWR) Design." July 1994.[75]

9. NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design." July 1994. Section 15.2, "Trip of All Reactor Internal Pumps and Pressure Regulator Down-Scale Failure."[76]

10. NUREG-1462, "Final Safety Evaluation Report (FSER) Related to the Certification of the Combustion Engineering (CE) System 80+ Design." August 1994.[77]

11~~2~~. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Overpressure," American Society of Mechanical Engineers.

~~3. Standard Review Plan Section 4.2, "Fuel System Design."~~[78]

~~4. Title 10 Code of Federal Regulations, Part 50, Revised January 1, 1980.~~[79]

~~5~~12. "Standard Safety Analysis Report--BWR/6," General Electric Company, April 1973.

~~6~~13. "Reference Safety Analysis Report--RESAR-3," Westinghouse Nuclear Energy Systems, November 1973; and "Reference Safety Analysis Report--RESAR-41," Westinghouse Nuclear Energy Systems, December 1973. "Reference Safety Analysis Report--RESAR-3S," Westinghouse Nuclear Energy Systems, July 1975; and "Reference Safety Analysis Report--RESAR-414," Westinghouse Nuclear Energy Systems, October 1976.

~~7~~14. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973.

~~8~~15. "Standard Nuclear Steam System B-SAR-205," Babcock & Wilcox Company, February 1974.

~~9~~16. ANSI N18.2,[80] "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American National Standards Institute (1974).

~~10~~17. ANS Trial Use Standard N212,[81] "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," American Nuclear Society (1974).

~~11. General Design Criterion 10, "Reactor Design."~~

~~12. General Design Criterion 15, "Reactor Coolant System Design,"~~

13. ~~General Design Criterion 26, "Reactivity Control System Redundancy and Capability."~~

14. ~~Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."~~

15. ~~Regulatory Guide 1.105, "Instrument Spans and Setpoints."~~[82]

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Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	Current PRB abbreviation	Change PRB to SRXB.
2.	Editorial revision	This definition appears much later in the text in the current version of the SRP section. It was moved forward to the first usage of "moderate frequency." The definition was edited because Reference 9 does not use the term "moderate frequency."
3.	Editorial revision	Wording was revised to clarify that this review is to ensure that there will be no fuel damage from specified moderate frequency events. The three General Design Criteria cited as acceptance criteria do not allow fuel damage.
4.	Editorial revision	Defined "SRP" as "Standard Review Plan."
5.	Editorial revision	Deleted "these types of" for simplicity. It is not clear what "these types of moderate frequency events" referred to.
6.	SRP-UDP format item	Changed Standard Format (which is imprecise) to Regulatory Guide 1.70 and eliminated an obvious reference.
7.	Editorial revision	Defined PWR.
8.	Current PRB designation	Changed PRB to SRXB in this sentence and the next.
9.	Current PRB designation	Changed PRB to SRXB.
10.	Current PRB designation	Changed PRB to SRXB.
11.	Editorial revision	Changed "assure" to "ensure."
12.	SRP-UDP format item	Added "Review Interfaces" to AREAS OF REVIEW.
13.	Current PRB designation	Changed PRB to SRXB.
14.	Editorial revision	Broke the current paragraph up into separate items for each interface.
15.	Current review branch designation	Changed PRB to HICB.
16.	SRP-UDP format item	Moved this paragraph down in the text. This review is now the responsibility of the SRXB. The Core Performance Branch is gone. The text was relocated to separate it from other branch reviews.
17.	Editorial revision	Eliminated this Review Interface because the review should conclude that the accidents considered will not result in fuel damage other than, perhaps, clad perforation.

Item	Source	Description
18.	SRP-UDP format item	Changed review branch to TSB.
19.	SRP-UDP format item	This paragraph appeared earlier in the text in the current SRP section. It was moved here to separate it from those review interfaces which are the responsibility of other branches.
20.	Current PRB designation	Changed PRB to SRXB.
21.	Editorial revision	Provided "GDC 10" as an initialism for General Design Criterion 10.
22.	Editorial revision	Changed "assure" to "ensure."
23.	Editorial revision	Provided "GDC 15" as an initialism for General Design Criterion 15.
24.	Editorial revision	Changed "assure" to "ensure."
25.	Editorial revision	Changed "breeched" to "breached."
26.	**Integrated Impact No. 1488**	Added GDC 17 as a new acceptance criterion, item C and renumbered next criterion accordingly.
27.	Editorial revision	Provided "GDC 26" as an initialism for General Design Criterion 26.
28.	Editorial revision	Changed "assure" to "ensure."
29.	Editorial revision	Changed "assuring" to "ensuring."
30.	Editorial revision	Changed "are" to "is." The subject of the clause is margin.
31.	Editorial revision	Changed "GDC" to "General Design Criteria" to accommodate plural usage (global change for this section).
32.	Editorial revision & SRP-UDP format item	Corrected "valves" to "values." Entered a citation for the ASME Code and eliminated the now-obvious reference.
33.	Editorial revision	Defined DNBR at its first point of use.
34.	Editorial revision	Defined CPR at its first point of use.
35.	Editorial revision	Defined MCPR at its first point of use.
36.	Editorial revision	Eliminated the clause about a radiological dose calculation. It is unacceptable that an event of moderate frequency in combination with a single failure result in significant fuel failure.
37.	Editorial revision	Corrected "valves" to "values."
38.	SRP-UDP format item	Updated the title of Regulatory Guide 1.105 to the title as it appears in Revision 2, February 1986.

Item	Source	Description
39.	SRP-UDP format items	Inserted the title of Regulatory Guide 1.53 since the title of RG 1.105 is included in the preceding item. Eliminated obvious reference. Note that Regulatory Guide 1.53 endorses two outdated standards: IEEE Std 279-1971 and IEEE Std 379-1972. There is a 1991 version of 279 and a 1988 version of 379.
40.	**Integrated Impact No. 795 and 796**	Added the ABWR and System 80+ SARs as references providing models approved for transient analyses.
41.	SRP-UDP format item	Corrected the abbreviation of the responsible branch, HICB. Added a clause identifying the SRPs covering the review against Regulatory Guide 1.105.
42.	SRP-UDP format item	Added "Technical Rationale" and lead-in paragraph to ACCEPTANCE CRITERIA.
43.	SRP-UDP format item	Provided technical rationale for GDC 10.
44.	SRP-UDP format item	Provided technical rationale for GDC 15.
45.	**Integrated Impact No. 1488**	Added Technical Rationale for GDC 17.
46.	SRP-UDP format item	Provided technical rationale for GDC 26.
47.	Current PRB designation	Changed PRB to SRXB.
48.	Editorial revision	Changed "are" to "is."
49.	Current PRB designation	Changed PRB to SRXB.
50.	Current review branch designation	Changed review branch to HICB.
51.	Current PRB designation	Changed PRB to SRXB.
52.	Current review branch designation	Changed review branch to HICB.
53.	Current PRB designation	Changed PRB to SRXB.
54.	**Integrated Impact No. 1488**	Added the new staff position from the CE 80+ FSER that indicates that LOOP may not be considered a single failure.
55.	Current PRB designation	Changed PRB to SRXB.
56.	Current PRB designation	Changed PRB to SRXB.
57.	Editorial revision	Modified to eliminate gender-specific reference.
58.	Editorial revision	Modified to eliminate gender-specific reference.
59.	SRP-UDP format item	Indicated that this review is now the responsibility of the PRB.
60.	Editorial revision	Corrected usage from "compared to" to "compared with" (global for this section).

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
61.	Editorial revision	Defined DNBR at point of its first use in the document. It need not be defined here.
62.	Editorial revision	The parenthesis is opened after "coolant conditions." The parenthesis is not closed in the current version. I believe the parenthetical information ends here.
63.	**Integrated Impact No. 1489**	Added a paragraph discussing the special criteria applied to a transient unique to the ABWR.
64.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
65.	Editorial revision	Modified to eliminate gender-specific reference.
66.	**Integrated Impact No. 1488**	Added GDC 17 to the list of acceptance criteria addressed in sample evaluation findings.
67.	**Integrated Impact No. 1488**	Added GDC 17 to the list of acceptance criteria addressed in sample evaluation findings.
68.	**Integrated Impact 1488**	Added GDC 17 to the list of acceptance criteria addressed in sample evaluation findings.
69.	SRP-UDP Format Item, Implement 10 CFR 52 Related Changes	To address design certification reviews a new paragraph was added to the end of the Evaluation Findings. This paragraph addresses design certification specific items including ITAAC, DAC, site interface requirements, and combined license action items.
70.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
71.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
72.	**Integrated Impact No. 1488**	Added GDC 17 to the list of references and renumbered other references accordingly.
73.	Editorial	Moved reference to RG 1.53 to be consistent with the ordering of references in other sections.
74.	SRP-UDP format item	Moved the reference to be consistent with other sections and corrected the title of Regulatory Guide 1.105 to the current title.
75.	**Integrated Impact No. 795**	Added a reference to the ABWR FSER as the source of the staff-approved model for transient analyses for the ABWR.
76.	**Integrated Impact No. 1489**	Added reference to the ABWR FSER, section 15.2 to support unique staff position for ABWR downscale failure of pressure regulator transient.

Item	Source	Description
77.	**Integrated Impact No. 796**	Added a reference to the CE System 80+ FSER as the source of the staff-approved model for transient analyses for the CE System 80+ reactor.
78.	Editorial revision	Eliminated the reference to another section of the SRP. Renumbered subsequent references.
79.	Editorial revision	Listed each GDC as a separate reference to make this section consistent with the other SRP sections.
80.	**Integrated Impact No. 686**	This standard may need to be updated to reflect the current version.
81.	**Integrated Impact No. 686**	This standard may need to be updated to reflect the current version.
82.	SRP-UDP format item	Re-ordered references to be consistent with other SRP sections.

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SRP Draft Section 15.5.1

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
686	Standards may need to be updated to reflect their current versions.	VI
795	Add analytical models approved for the ABWR.	II and VI
796	Add analytical models approved for the CE System 80+.	II and VI
1488	Modified the Acceptance Criteria to include GDC 17 and revised the Review Procedures to incorporate staff guidance regarding the assumption of LOOP, in addition to a limiting single failure event, for the analysis of reactor coolant pump rotor seizure and reactor coolant pump shaft breaks.	II, III, IV, and VI
1489	Added a discussion regarding a unique acceptance criterion for the ABWR's postulated downscale failure of pressure regulators.	III and VI